**Third United Nations International Conference on the Peaceful Uses of Atomic Energy**

A/CONF.28/A/543
AUSTRALIA
February 1964
Original: ENGLISH

Confidential until official release during Conference

**Abstract**

# THE DEVELOPMENT OF BERYLLIA-BASED FUELS FOR H.T.G.C. REACTOR SYSTEMS

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As part of a High Temperature Gas Cooled Reactor feasibility study by the Australian Atomic Energy Commission (Alder and Roberts - this Conference), a materials research programme has been in progress since 1958 to evaluate the potential of beryllium oxide as a fuel diluent in dispersion fuels and also as a moderator material. The programme has been designed around an all-ceramic core, cooled by carbon dioxide, employing highly rated, fission-product-retentive, high burn-up fuels with a maximum fuel surface temperature of 1000°C. Although various fuel cycles are being considered, experimental work on materials has been confined mainly to a $U^{235}$/Th cycle and dispersion fuels consisting of homogeneous fuel particles of $(U\ Th)O_2$ in a matrix of BeO. This work has involved fuels with from 3-30 vol % of irregular or spherical fuel particles ranging in size from 10-200 microns.

The paper surveys the Australian work directed towards understanding and evaluating the behaviour of beryllia and beryllia-based fuels in relation to the conditions under which they might be used in the proposed reactor system. Attention is confined mainly to the problems of resistance to thermal stress, fission-product retention, corrosion resistance, irradiation behaviour, chemical reprocessing and overall fuel cycle costs and their relation to fabrication variables, microstructure, temperature and neutron environment.

Irradiation experiments in essentially thermal fluxes have shown that stability under fission-product damage is excellent up to $2 \times 10^{21}$ fissions/$cm^3$ of fuel particle at temperatures from 600-850°C with thermal stresses up to 30 000 psi.

A more serious problem is fast-neutron damage to the beryllia matrix which causes powdering at high doses (Elston and Caillat, 2nd UN Geneva Conference, Paper 15/P/1159). At temperatures up to 800°C this has been

shown to be due to displaced atoms causing anisotropic growth of crystals, high intergranular stresses and eventually microcracking. Microcracking in the fuel matrix must be avoided since it causes loss of strength, reduction in thermal conductivity and increased release of fission-products. The onset of microcracking is shown to be dependent on material structure, in particular the grain-size, as well as the irradiation temperature, dose and dose-rate. On the basis of the results, analytical expressions have been developed to allow predictions of the property changes to be expected in beryllia under a variety of reactor conditions. Fine grained beryllia (grain size less than 5 microns) should withstand a dose of $2 \times 10^{21}$ nvt (fast) at 400°C and greater than $10^{22}$ nvt (fast) at 800°C and above; this should allow beryllia to be used satisfactorily as a fuel diluent in dispersion fuels, but not as fixed moderator to last the reactor life.

Fabrication techniques for obtaining high-density fine-grained bodies with adequate resistance to thermal stress and satisfactory fission-product retention are reported together with possible techniques for chemical re-processing of the fuel. Corrosion losses at temperatures up to 1000°C due to water vapour in the coolant are shown to be almost negligible.

The preliminary work has given confidence in the technical performance of all-ceramic beryllia-based fuel provided it is used in the form of simple shapes which facilitate fabrication and handling problems and which suit the intrinsically brittle nature of the material. A progress report is given of further work directed towards a more complete engineering evaluation of spherical-shaped fuel elements for a pebble-bed reactor concept.